May 21, 1999

Mr. R. P. Powers
Senior Vice President
Nuclear Generation Group
American Electric Power Company
500 Circle Drive
Buchanan, MI 49107-1395

SUBJECT: NRC INSPECTION REPORT 50-315/99009(DRS); 50-316/99009(DRS)

Dear Mr. Powers:

On April 23, 1999, the NRC completed a special inspection conducted at your Buchanan Michigan Corporate facility. This inspection was an examination of activities under your license as they relate to your implementation of the Expanded System Readiness Review (ESRR) program at your D. C. Cook Units 1 and 2 reactor facilities. The NRC understands that these reviews are intended to provide assurance that safety-related plant systems fulfill their design basis safety functions and to determine system restart readiness. Effective implementation of the ESRR process will address portions of Case Specific Checklist Items No. 3, “Programmatic Breakdown in the Maintenance of the Design Basis,” and No. 13, “Systems and Containment Readiness Assessment,” that were established through the NRC’s Manual Chapter 0350, “Staff Guidelines for Restart Approval.” The NRC will continue to monitor and assess the effectiveness of the problem identification phase of your system readiness evaluations. The enclosed report documents the results of the inspection.

Overall, the implementation of the ESRR process observed was considered effective. The inspection focused on the reactor coolant and the control room instrumentation distribution system assessments. For these systems, the scope of the planned review areas were broad and generally consistent with the purpose of the review effort to confirm the performance of system safety functions. Further, the breadth and depth of material reviewed for these system assessments was appropriate and ESRR teams were effective at identification of substantive design issues potentially impacting system safety functions. However, the failure of the reactor coolant ESRR team to identify nonconservative assumptions for reactor coolant system inventory in station blackout calculations demonstrated a lapse in the technical rigor for this assessment and indicated a need for more focus on the quality of the review effort. Additionally, inspectors were concerned that lack of control of indices of vendor analysis could adversely impact the ESRR team assessments which rely on the material in these index lists to identify the up to date source materials for the system reviews.

The system readiness review board (SRRB) was observed to be following a structured review process that offered a consistent approach for effectively reviewing the large amount of information contained in a typical ESRR system report. The Auxiliary Building Ventilation

System ESRR report presentation demonstrated the ESRR team manager's thorough understanding of the system under review and of the relative significance of the system findings and assessments. However, the inspectors were concerned that performing significant additional licensing basis reviews (e.g., review of the historical design and licensing basis notebooks) following the SRRB approval process could change the system assessments and serve to "bypass" the SRRB review process step. Your staff's prompt actions to suspended the systems scheduled for SRRB review and approval pending the receipt and review of the historical design and licensing basis notebooks by the ESRR teams demonstrated an effective response to this concern.

Based on the results of this inspection, the NRC has determined that three violations of NRC requirements occurred. The first violation pertained to the Unit 1 tube plugging and sleeving activities which have occurred outside a formal design control process and represented a design control process bypass. The nonproceduralized process used for these activities did serve to accomplish some of the key functions of a formal design control process such as safety evaluations, analysis and technical specification changes. However the processes used were not effective in review of the vendor analysis, use of plant drawing controls or consideration of vendor specified testing. This violation appears to be another manifestation of the design control breakdown that contributed to the extended shutdown and improvement initiatives under way at D. C. Cook.

The second and third violation pertained to inadequate operability determination screenings of safety related equipment that indicate a lapse in rigor for the performance of operability determination screenings. Further, the inspectors identified several examples of operability determinations that contained poor quality engineering assessments and a large backlog (in excess of 500 items) of operability questions which have not received timely engineering resolution. These issues are of concern as they demonstrate poor engineering support for the operability determination process.

These violations are being treated as Non-Cited Violations (NCVs), consistent with Appendix C of the Enforcement Policy. These NCVs are described in the subject inspection report. If you contest the violation or severity level of these NCVs , you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with copies to the Regional Administrator, Region III, and the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

Appendix C of the Enforcement Policy requires that for Severity Level IV violations to be dispositioned as NCVs, they be appropriately placed in a licensee corrective action program. Implicit in that requirement is that the corrective action program be fully acceptable. The D. C. Cook Plant corrective action program was not adequate and has been the focus of significant attention by your staff to improve the program. While your staff and the NRC have not yet concluded that the corrective action program is fully effective, the corrective action and design control program improvement efforts are underway and captured in the D. C. Cook Plant Restart

Plan which is under the formal oversight of the NRC through the NRC Manual Chapter 0350 process, “Staff Guidelines for Restart Approval.” Consequently, these issues will be dispositioned as NCVs.

In accordance with 10 CFR 2.790 of the NRC'S "Rules of Practice," a copy of this letter, the enclosure, and your response to this letter, if you choose to provide one, will be placed in the NRC Public Document Room.

We will gladly discuss any questions you have concerning this inspection.

Sincerely,

Original /s/ J. A. Grobe

John A. Grobe, Director
Division of Reactor Safety

Docket Nos. 50-315; 316
License Nos. DPR-58; DPR-74

Enclosure: Inspection Report 50-315/99009(DRS);
50-316/99009(DRS)

cc w/encl: A. C. Bakken III, Site Vice President
M. Rencheck, Vice President, Nuclear Engineering
R. Whale, Michigan Public Service Commission
Michigan Department of Environmental Quality
Emergency Management Division
MI Department of State Police
D. Lochbaum, Union of Concerned Scientists

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-315; 50-316
License Nos:	DPR-58; DPR-74
Report No:	50-315/99009(DRS); 50-316/99009(DRS)
Licensee:	Indiana Michigan Power Company
Facility:	Donald C. Cook Nuclear Generating Plant
Location:	American Electric Power Corporate Office Buchanan, Michigan
Dates:	April 5-23, 1999
Inspectors:	R. Mendez, Reactor Engineer M. Holmberg, Reactor Engineer N. Shah, Reactor Engineer
Approved by:	John Jacobson, Chief Mechanical Engineering Branch Division of Reactor Safety

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EXECUTIVE SUMMARY

D. C. Cook, Units 1 and 2
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This was an engineering special inspection primarily to review the expanded system readiness review program. Specifically, this inspection focused on: (1) the scope of the expanded system readiness reviews and the breadth, depth and quality of the system assessments that serve to identify issues potentially impacting system safety functions; (2) the control processes controls applied for past steam generator tube plugging and sleeving activities; and (3) the engineering support for the operability determination process.

- The inspectors considered the scope of the planned review areas for the reactor coolant system to be broad and consistent with the purpose of the review effort to confirm the performance of system safety functions (Section E1.1).

- The breadth and depth of material reviewed for the reactor coolant system assessments was generally appropriate and the reactor coolant system review team was effective at identification of substantive design issues potentially impacting system safety functions. However, the failure of this team to identify nonconservative assumptions for station blackout reactor coolant inventory calculations demonstrated a lapse in the technical rigor for this assessment and indicated a need for more focus on the quality of the review effort (Section E1.1).

- The inspectors were concerned that lack of control of indices of vendor analysis could adversely impact the expanded system readiness review team assessments which rely on the material in these index lists to identify the up-to-date source review materials (Section E1.1).

- The control room instrumentation distribution system assessments were thorough and focused on design attributes and the control room instrumentation distribution system review team identified a number of significant design related issues. However, the lack of receipt/review of historical design and licensing data (deliverable 2 notebooks) had a potential generic impact on the quality of the ongoing expanded system readiness review team assessments (Section E1.2).

- The inspectors identified that Unit 1 steam generator tube plugging and sleeving activities occurred outside a formal design control process and represented a design control bypass which was considered a Non-Cited Violation of 10 CFR 50 Appendix B, Criterion III. The licensee processes used to review the vendor analysis were not effective in translating design related information into the plant design basis documents, nor were plant drawing controls utilized, and a vendor specified leak test was not performed and/or adequately considered (Section E1.3).

- The inspectors identified that a change in the design fouling factor for the Unit 1 steam generator U-tubes had never been formally reviewed/ evaluated by the licensee nor incorporated into the updated final safety evaluation report (Section E1.4).

- The inspectors identified several examples of operability determinations that contained poor quality engineering assessments. These examples combined with the large backlog (in excess of 500 items) of operability questions which have not received timely engineering resolution, demonstrated poor engineering support for the operability determination process (Section E2.1).

- In general the operability determination screenings reviewed by the inspector were adequate, however, the inspectors identified two examples where operability determination screenings of safety related equipment had not been adequately performed. These issues were considered examples of Non-Cited Violations of 10 CFR 50 Appendix B, Criterion V and indicated a lapse in rigor for the performance of operability determination screenings (Section E2.2).

- The system indexed data base system represented a powerful resource tool that enabled the expanded system readiness review teams to review an single integrated list of system deficiencies, work or modifications and as such, represented an essential element for effective system reviews. However, the expanded system readiness review teams were being adversely affected by the ongoing problems created by the disparity of system designators used in system indexed data base system and other plant data bases (Section E3.1).

- The system readiness review board members followed a structured review process that offered a consistent approach for effectively reviewing the large amount of information contained in a typical expanded system readiness review system report. The auxiliary building ventilation systems expanded system readiness review report presentation demonstrated the team manager’s thorough understanding of the system under review and of the relative significance of the system findings and assessments (Section E7.1).

- The inspectors were concerned that performing significant additional licensing basis reviews (e.g. review of the historical design and licensing basis notebooks) following the system readiness review board approval process could change the system assessments and serve to “bypass” the system readiness review board review process step. The licensee subsequently suspended the systems scheduled for system readiness review board review and approval pending the receipt and review of the historical design and licensing basis notebooks by the expanded system readiness review teams. Inspectors considered this action to demonstrate a prompt effective management response to this concern (Section E7.1).

Report Details

III. Engineering

E1 Conduct of Engineering

E1.1 Reactor Coolant System (RCS) Expanded System Readiness Review (ESRR) Assessments

a. Inspection Scope (40500, 37550, 37700, 93809)

Inspectors reviewed 20 of the 149 assessments planned for the RCS and RCS pressure relief systems, which had been completed through the RCS ESRR team manager review.

b. Observations and Findings

The scope of the RCS ESRR effort was defined in the system assessment matrix approved by the System Readiness Review Board (SRRB) on March 12, 1999. This matrix defined 13 system attributes (included items such as Pressure/Fission Product Boundary, Transfer Heat, Reactor Vessel and Vents) that were reviewed in five major topical areas (Design Basis, Licensing Basis Documents, Operations, Maintenance, Surveillance, Physical Plant, and Programs/Processes). The scope of the areas for review was generally broad and consistent with the purpose of the review effort to confirm the performance of system safety functions. However, the review scope did not include planned assessments of the RCS weld inspections conducted in accordance with Section XI of the American Society of Mechanical Engineers (ASME) Code. Because these weld inspections provide assurance of continued RCS boundary integrity, the lack of a review/assessment in this area was inconsistent with a completely comprehensive effort to validate the principal RCS safety function as a barrier to the release of reactor coolant and radioactive materials.

The RCS assessments were effective at identification of substantive design related issues. The ESRR team identified that no basis existed for the capacity of the pressurizer safety relief valves and that the low temperature over-pressure protection system (LTOP) setpoints were based on an incorrect assumption of RCS volume based on a nonconservative input for the number of Unit 2 steam generator tubes. Inspectors evaluated the information sources (calculations, drawings, vendor analysis, etc) reviewed by the ESRR teams for each of the individual assessments which had been completed through the team leader review. The inspectors found that these individual assessments had been based on reviews of the appropriate data sources. One exception existed for the Unit 1 steam generator tube plugging assessment DWEA066 “Transfer Heat” which did not contain the steam generator general design specification G-677164 “Reactor Coolant System Series “51" Steam Generator.” This assessment was undergoing team leader review, but the preparer of this assessment acknowledged that the steam generator design specification should have been included for a more complete and comprehensive review. The depth of the reviews for these assessments generally stopped at the vender supplied analysis level (e.g., Westinghouse WCAP document). The team manager stated that the input assumptions to many of these vender analysis would be investigated by the team during the planned review and assessment of the accident analysis. In general, the inspectors considered the breadth and depth of material reviewed to be appropriate and effective at identification of design issues potentially impacting system safety functions.

Inspectors identified that Calculation TH-97-08 “Reactor Coolant System Inventory After Station Blackout,” Revision 0, used nonconservative assumptions for RCS inventory derived from Calculation TH-89-08 “Reactor Coolant System Inventory After Station Blackout,” Revision 0. Both of these calculations had been reviewed and accepted by the RCS team in assessment DWEA033 “Required Reactor Coolant Inventory to Support Station Blackout.” These calculations nonconservatively assumed a primary volume associated with steam generators which had not undergone tube plugging. Inspectors calculated the magnitude of the input error was approximately 900 cubic feet of RCS inventory assuming the 30 percent tube plugging allowed under the Unit 1 plant licensing basis or 440 cubic feet at the current tube plugging level of 16 percent. Due to relatively large margin of RCS inventory available (3000 cubic feet) this error did not change the overall conclusion for this calculation (e.g., that the core would remain covered). However, the failure of the RCS team to identify this issue demonstrated a lapse in the technical rigor for this assessment and indicated a need for more focus on the quality of the review effort. The inspectors noted that a licensee review effort conducted as part of the Unit 1 steam generator replacement project identified the nonconservatisms in these calculation assumptions (condition report 99-8063) independent of and coincidental with the inspectors’ review.

The RCS team was effectively utilizing industry experience to identify potential design problems. The industry had experienced degradation (blistering) of pressurizer relief tank (PRT) internal coating (Americoat 55) at another reactor plant (H.B. Robinson) which had been identified through inspections of this tank. Degradation of this coating could result in tank corrosion and failure. The team identified that the D.C. Cook PRT internal coating was Americoat 55 which had not been inspected over the life the plant and was not in a periodic inspection program. This issue demonstrated effective application of industry experience and the team documented this potential problem in condition report 99-3453. However, the inspectors’ questions pertaining to the service temperature range and life of the PRT tank coating prompted the team to more fully develop this issue. Specifically, the team’s continued investigation identified that the tank coating was not qualified for the full design temperature range of the tank and that the coating was not qualified for acidic environments. This issue illustrated an initial lack of follow up on the identified issue and again demonstrated a need to focus on the quality of the review effort.

The RCS ESRR team relied on an index of vendor analysis (e.g., Westinghouse WCAPs) to identify the appropriate review material to support the assessments performed. The RCS team identified that this list did not include WCAP-12245-R3-AD3 “Addendum 3 To Steam Generator Tube Plug Integrity Summary Report.” This list identified only the Addendum 2 version of WCAP-12245-R3-AD2. Inspectors’ questions prompted the licensee staff to identify in condition report 99-09108 that the WCAP index did not reflect the latest information. The licensee documented that the inaccuracy of this particular index did not adversely effect the ESRR effort. However, the inspectors noted that the latest version of WCAP-12245-R3-AD3 was in fact used to support the RCS assessments for tube plugging. Because the licensee currently had no procedural guidance or control for the maintenance of indices of vendor analysis, the inspectors were concerned that this issue could adversely impact the ESRR team assessments. Specifically, the ESRR teams rely on the material in these index lists to identify the up-to-date source review materials such that effective system reviews are performed. The nuclear records management personnel indicated that actions were being considered to develop procedural guidance for the control of indices of vendor analysis.

c. Conclusions

The inspectors considered the scope of the planned review areas for the RCS to be broad and consistent with the purpose of the review effort to confirm the performance of system safety functions. One exception existed in that the system review scope did not include planned assessments for the RCS weld inspections conducted in accordance with the ASME Code. Because these weld inspections provide assurance of continued RCS boundary integrity, the failure to include a review/assessment in this area was inconsistent with a complete and comprehensive effort to validate the principal RCS safety function.

The breadth and depth of material reviewed for the RCS system assessments was generally appropriate and RCS review team was effective at identification of substantive design issues potentially impacting system safety functions. However, the failure of the RCS team to identify nonconservative assumptions for RCS inventory station blackout calculations demonstrated a lapse in the technical rigor for this assessment and indicated a need for more focus on the quality of the review effort. Additionally, inspectors were concerned that lack of control of indices of vendor analysis could adversely impact the ESRR team assessments which rely on the material in these index lists to identify the up-to-date source review materials.

E1.2 120 VAC Vital Buses and Control Room Instrumentation Distribution (CRID) Inverter ESRR Assessments

a. Inspection Scope

The inspectors reviewed 21 of the 59 assessments planned for the 120 VAC vital buses and CRID inverter. These assessments were completed through the CRID ESRR team manager review.

b. Observations and Findings

The scope of the CRID ESRR reviews was defined in the system assessment matrix that was approved by the SRRB on March 5, 1999. The matrix assessment included evaluation of system mitigation functions (attributes) against plant configuration design, licensing, operations and maintenance documents. This scope included design attributes for review such as the CRID input sources, the CRID output requirements, inverter requirements, isolimeter requirements, power source transfer capabilities and electrical coordination. Other design attributes were also defined that were common to other systems such as station blackout operation, high energy line break impact on the 120 VAC vital bus systems and seismic qualification of the CRID related to operability of the engineered safety features systems. The individual ESRR assessments identified a number of significant issues such as the lack of electrical load and voltage calculations to support CRID output requirements, incorrect fuses installed in the field and numerous preventative maintenance tasks that were either not performed or were overdue. The completed assessments were in general thorough and focused on appropriate design attributes.

The inspectors noted that the CRID team verified the design information in the UFSAR but did not review the design information in the original FSAR. As a consequence the CRID team had not identified system design and configuration discrepancies that existed between the UFSAR and FSAR until prompted by inspectors' questions as discussed below. The inspectors were concerned that this condition was generic, because none of the ESRR teams had received the historical design and licensing information contained in the "deliverable 2" system notebooks.

During review of the CRID system the inspectors noted inconsistent information between the FSAR, UFSAR and the vendor technical manual regarding the balance-of-plant isolimeter. The UFSAR stated that the inverter may derive its input from the balance-of-plant regulating transformer (isolimeter) whose input is a 600 VAC essential service supply source. The FSAR stated that the output of the isolimeter was regulated to provide 120 VAC plus or minus three percent. However, the inspector noted that the vendor manual specified that the input voltage for the isolimeter was 575 VAC and the output voltage was 120 VAC but the vendor manual did not specify a tolerance for the input voltage or the output voltage regulation. The team indicated they would issue a condition report and determine what the appropriate design input and output tolerances should be for the isolimeter.

The inspectors noted a difference in the system description of the 120 VAC system between the FSAR and the UFSAR. The FSAR described the inverter cabinet as receiving its inputs from two AC sources. These sources were from the output of a battery charger whose input is a 600 VAC source and the output of a rectifier whose input is a 600 VAC source independent of the battery charger source. However, the current UFSAR listed the input to the inverter from a single 600 VAC supply to the battery charger. Additionally, the FSAR listed the inverter voltage as being automatically regulated at 118 VAC with plus or minus three percent voltage tolerance. While the UFSAR listed the inverter voltage as being regulated automatically at 118 VAC but with a plus or minus two percent voltage tolerance.

In discussing this issue with the CRID team, the team indicated that the rectifier, as discussed in the FSAR, was installed inside the inverter and therefore, it was likely the rectifier was a safety-related component. The present plant configuration had an isolimeter that was purchased non-seismically qualified and consequently, the isolimeter was classified as non-safety-related. The CRID team committed to review the modification that changed the AC rectifier supply to the non-safety-related isolimeter and to review the output AC voltage regulation differences between the FSAR and the UFSAR.

c. Conclusions

The CRID system assessments were thorough and focused on design attributes and the CRID review team identified a number of significant design related issues. However, inconsistencies in the design voltage tolerances for the isolimeter and safety related power supplies not been identified by this team, in part, due to the lack of a review of the original FSAR. This oversight was attributed to the lack of receipt of historical design and licensing data (deliverable 2 notebooks). The lack of receipt/review of deliverable 2 notebooks had a potential generic impact on the quality of the ongoing ESRR team assessments.

E1.3 Steam Generator Tube Plugging Activity Bypasses the Design Control Process

a. Inspection Scope (40500, 37550, 37700, 93809)

Inspectors interviewed personnel, reviewed safety evaluations, technical specification amendments and other design basis analysis pertaining to steam generator tube plugging and sleeving operations which have typically occurred during the Unit 1 and 2 refueling outages over the life of the plant.

b. Observations and Findings

The inspectors identified that the plant modification process had never been implemented for the plugging and sleeving of defective steam generator tubes which had typically occurred during refueling outages over the life of the plant. An average of sixteen percent of the Unit 1 steam generator tubes were plugged as a result of past plugging and sleeving activities. Unit 2 steam generators had been replaced in 1988 and had less than one percent of the tubes plugged, which was within the Unit 2 replacement steam generator design specifications. The licensee had completed Unit 1 plugging and sleeving activities under job order and work packages which referenced vendor procedures. The inspectors were concerned that this activity constituted a bypass of the existing design control process and the licensee documented this issue in condition report 99-06774. The inspectors identified this issue by performing a review of RCS modifications documented in design control packages, minor modifications and field changes. The RCS ESRR team reportedly had not completed a detailed review of plant modifications and ESRR project managers concluded that the RCS ESRR team would have likely identified this issue prior to completion of the RCS review efforts. The inspectors could not confirm the validity of this conclusion.

The inspectors and the investigation in progress by the design engineering organization confirmed that a nonproceduralized process was in place and did serve to accomplish some of the key functions of a formal design control process. For example safety evaluations were completed for the tube plugs and sleeves installed. Supporting vendor analysis and bounding accident analysis/evaluations were performed. Additionally, the licensee had implemented appropriate changes to plant technical specifications where required. However, the inspector identified several examples discussed below where the process used was not effective in implementing design control elements.

The licensee had previously identified in condition report 98-1655, dated March 17, 1998, that existing engineering procedures were deficient with respect to implementing ANSI N45.2.11-1974 Section 5.4 requirements for the design interface transmittals of information (such as with Westinghouse). The inspector confirmed that the licensee processes used to review the vendor analysis associated with tube plugging activities were not effective in translating design related information into the plant design basis documents. For example, the licensee accepted WCAP-14286, "American Electric Power Service Corporation Donald C. Cook Nuclear Plant Unit 1 Steam Generator Tube Plugging Program Engineering Report," and WCAP-14285 "Donald C. Cook Nuclear Plant Unit 1 Steam Generator Tube Plugging Program Licensing Report, " Revision 1, as the design and licensing basis for allowing up to 30 percent of the tubes to be plugged in the Unit 1 steam generators. These analysis were reviewed by the licensee, however, lack of standardized guidance or direction for these reviews resulted in design data that was not adequately translated into other design basis documents. Specifically, the revised reactor coolant system volumes created by the increased plugging of steam generator tubes was not translated in station blackout inventory calculations (Section E1.1). A change to the Unit 1 steam generator tube fouling factor supporting the analysis conclusions did not receive licensee review nor was the new revised value updated in the UFSAR (Section E1.4). The changes to the Unit 1 thermal hydraulic design data were not incorporated into the original thermal hydraulic report WTD-PWEE 70-31, "51 Series Steam Generator Thermal Hydraulic Data Report For Donald C. Cook Power Station 1 & 2," as called out by paragraphs 1.3, 1.4.1 and 1.4.2 of the steam generator general design specification G-677164 "Reactor Coolant System Series "51" Steam Generator." Additionally the nuclear records management system did not maintain adequate controls for the indices of vendor analysis (Section E.1.1).

The licensee maintained the status of the Unit 1 steam generator tubes and sleeves which had been installed on a figure/drawing depicting the location of the plugs or sleeves in terms of tube sheet row and column location. This drawing was reportedly being updated each outage, however it did not have a number and was not entered into the licensee's formal drawing control program. Therefore, inspectors were concerned that revisions and updates to this drawing were not formally controlled, increasing the chances for errors and omissions. Further, no proceduralized process existed to define responsibilities or record calculations to convert the number of sleeved tubes into an equivalent percentage plugging that was reportedly derived from information in WCAP -12623, "American Electric Power D.C. Cook Unit 1 Steam Generator Sleeving Report," (Mechanical Sleeves) dated June 1990.

The licensee had normally installed mechanical plugs during past outages in the Unit 1 steam generators, and on at least two occasions a welded plug had been installed. The inspectors identified that the licensee had not performed the pressure test of the steam generator tubes and plugs as specified by the vendor technical manual. Section 5.6 and 5.7 of the vendor technical manual for the Unit 1 steam generator VTD-WEST-0409 "Instructions for Vertical Steam Generators for American Electric Power Services Corporation Donald C. Cook Nuclear Plant Bridgman Michigan," specified conducting a 840 psig pressure test from the secondary side of the steam generator to identify leakage from the plugged or other in service tubes following plugging activities. The inspectors noted that the leakage test would put the face of the tube sheet in tension and present a more severe test for tube plug installation than minimum ASME Code and

Technical Specifications requirements. The inspectors noted that other utilities with nuclear power stations had conducted secondary side pressure testing following installation of welded steam generator plugs. As of April 23, 1999, the licensee had not completed a technical evaluation to reconcile the lack of a vendor technical manual specified post plug installation leakage test.

Section 4.2 of Procedure 227400-STG-5400-3 "Design Change Packages," Revision 1, defined a design change as a physical change to structures, systems or components that alters their technical or quality requirements. Additionally, 10 CFR 50, Appendix B, Criterion III, "Design Control," specified that design control measures shall be applied to items such as thermal, hydraulic and accident analysis. In this case, the steam generator plugging and sleeving activities constituted a design change in that, the physical change to a component (steam generator) impacted reactor coolant thermal, hydraulic and accident analysis. Based on the issues discussed above, the Unit 1 steam generator tube plugging and sleeving was not adequately controlled under a formal design control process such as that described by procedure 227400-STG-5400-3. The failure to implement adequate design controls for the Unit 1 steam generator tube plugging and sleeving activities that occurred during outages in March of 1997, September of 1995, April of 1994, August of 1992, November of 1990 and April of 1989 is a violation of 10 CFR 50, Appendix B, Criterion III. This Severity Level IV violation is being treated as a Non-Cited Violation (NCV). Appendix C of the Enforcement Policy requires that for Severity Level IV violations to be dispositioned as NCVs, they be appropriately placed in a licensee corrective action program. Implicit in that requirement is that the corrective action program be fully acceptable. The adequacy of D.C. Cook's corrective action program is of concern to both the NRC and Indiana Michigan Power. Because improving the corrective action program to a satisfactory status is an integral part of Indiana Michigan Power's "Restart Plan" and is under the formal oversight of the NRC through the NRC Manual Chapter 0350 Process, "Staff Guidelines for Restart Approval," this issue will be dispositioned as an NCV (NCV 50-315/99009-01(DRS)).

c. Conclusions

The inspectors identified that Unit 1 steam generator tube plugging and sleeving activities occurred outside a formal design control process and represented a design control bypass which was considered a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion III. The nonproceduralized process used by the licensee accomplished some of the key functions of a formal design control process such as safety evaluations, analysis and technical specification changes. However, the licensee processes used to review the vendor analysis were not effective in translating design related information into the plant design basis documents, nor were plant drawing controls utilized, and a vendor specified leak test was not performed and/or adequately considered.

The RCS review team reportedly had not completed a detailed review of plant modifications and ESRR project managers concluded that this team would have likely identified this issue prior to completion of the RCS review efforts. The inspectors could not confirm the validity of the licensee's conclusion.

E1.4 Unreviewed Change in Steam Generator Tube Fouling Factor

a. Inspection Scope (40500, 37550, 37700, 93809)

Inspectors reviewed vendor analysis and documentation supporting the increased tube plugging levels for the Unit 1 steam generators.

b. Observations and Findings

The inspectors identified that a change in the design fouling factor for the Unit 1 steam generator U-tubes had never been formally reviewed and evaluated by the licensee when the licensee accepted WCAP-14286 and WCAP-14285 in early 1997 as the design and licensing basis to establish the 30 percent tube plugging limit for the Unit 1 steam generators. Further, the change in the Unit 1 steam generator fouling factor had never been incorporated into Table 4.1-5 "Steam Generator Design Data" of the UFSAR. The inspectors considered this unreviewed fouling factor to be another unintended consequence resulting from the lack of adequate design control for steam generator tube plugging and sleeving activities discussed in Section E1.3.

Heat transfer resistance across the steam generator U-tubes is a design parameter established in the original steam generator design specification G-677164 "Reactor Coolant System Series "51" Steam Generator." The heat transfer resistance affects the ability to remove heat from the reactor coolant system and core. WCAP -14285 "Donald C. Cook Nuclear Plant Unit 1 Steam Generator Tube Plugging Program Licensing Report," Revision 1 Section 3.3.4.6 describes that a decrease in heat transfer ability associated with tube plugging aggravates the heat up portion for the loss of all AC power transient and increases the potential for filling the pressurizer. A similar affect would be expected for an increase in the steam generator fouling factor which would increase the overall heat transfer resistance.

The total heat transfer resistance across the steam generator U-tubes established in the Unit 1 steam generator design specification G-677164 is divided into four components; the inside heat transfer resistance, outside boiling film resistance, tube wall heat transfer resistance and fouling resistance. The original fouling resistance specified for the steam generators was 0.0002 FT^2-hr-^{o}F/BTU. This number represented 21 percent of the overall tube heat transfer resistence. This number was changed to 0.00005 FT^2-hr-^{o}F/BTU for Unit 2 following the replacement of steam generators in 1988. For Unit 1, the fouling factor was changed to this same value when the analysis WCAP -14285 to support the 30 percent tube plugging limit was accepted by the licensee in early 1997. The revised 0.00005 FT^2-hr-^{o}F/BTU value represented a decrease to 5.2 percent of the overall heat transfer resistance. The basis for the change in fouling factor was established in a Westinghouse letter SGPV-255 dated May 20, 1980. This document discussed that the 0.00005 FT^2-hr-^{o}F/BTU value represented a 95 percent probability that measured fouling factors would be bound by this number. However, the inspector identified that values of fouling factor derived from 1991 measured parameters for the Unit 1 Steam generators recorded in Westinghouse calculation PCWG-AEP-098 "PCWG Parameters for AEP: 30% SGTP Program," Revision 0 were greater than the bounding value and represented numbers as high as 12 percent of the total heat transfer resistance. Additionally, the 0.00005 FT^2-hr-^{o}F/BTU value was based on steam generators which had been in operation for an average of three years and no allowance for long term fouling was included in this number.

Therefore, inspectors were concerned that the change to a 0.00005 FT^2-hr-^oF/BTU value for the steam generator tube fouling factor may not be conservatively bounding for transients such as the loss of AC power or loss of feedwater. The inspectors' questions prompted discussions between the licensee and Westinghouse. Westinghouse provided the licensee an E-mail dated April 18, 1999 which provided their response to the inspectors' concern. In this response Westinghouse concluded that differences in transient response between the expected range of fouling factors has an insignificant effect. However, the response did not include reference to specific Westinghouse sensitivity studies or analysis demonstrating this conclusion. The licensee issued condition report 99-09156 to document the unreviewed change in the steam generator tube fouling factor. The inspectors considered this issue an inspection followup item (IFI 50-315/99009-02(DRS)) pending licensee review and resolution of this issue.

c. Conclusions

The inspectors identified that a change in the design fouling factor for the Unit 1 steam generator U-tubes had never been formally reviewed/ evaluated by the licensee nor incorporated into the UFSAR. The inspectors considered this unreviewed fouling factor to be another unintended consequence of the lack of adequate design control for steam generator tube plugging and sleeving activities discussed in Section E1.3.

E2 Engineering Support of Facilities and Equipment

E2.1 Poor Quality and Untimely Engineering Support of the Operability Process

a. Inspection Scope (40500 and 37550)

Inspectors reviewed a sample of operability determinations that required engineering support.

b. Observations and Findings

From a screening of condition reports in the System Index Database System (SIDS) the ESRR teams had documented problems with only two open operability determinations from a population of over 400 open operability determinations. The inspectors reviewed a sample of these operability determinations that required engineering support. The inspectors identified three operability evaluations with poor quality engineering documentation supporting the operability determination and a backlog of operability questions which have not received timely engineering resolution, that demonstrated poor engineering support for the operability process.

The inspectors identified two open operability determinations with weaknesses in the engineering technical basis and documentation supporting the operability evaluation. In condition report 99-2455 for Unit 2 issued on February 12, 1999 and condition report 99-2466 for Unit 1 issued on February 11, 1999, the licensee documented a concern for cavitation of the residual heat removal (RHR) system pumps. The licensee had taken compensatory actions to restrict flow rate and system lineups to minimize the vibration and potential for cavitation within the RHR system. The engineering department had also provided an assessment of the potential for fatigue cracking in the RHR system piping. This analysis concluded that the past average stresses resulting from RHR system vibrations were below the endurance fatigue limits for the materials used in the piping systems. This analysis applied to those structures systems and components of the system which had not experienced fatigue failures in the past and was based on the licensee's best guess at the number of predicted stress cycles. The inspector noted that this type of logic ignored the fatigue crack initiation and growth life cycle and was inadequate to conclude that vibratory stresses were below the fatigue endurance stress limit. Inspectors concluded that fatigue cracking in RHR system components could not be ruled out as a potential long-term failure mode. Additionally, the licensee originally planned to perform dye penetrant examinations of the RHR branch line welds. This action was not performed based on an unsupported engineering assessment that postulated fatigue cracks would occur from the inside of branch line welds and progress to the outside. The inspector considered these examples to demonstrate poor quality engineering assessments.

The inspector identified an open operability assessment which lacked documented engineering calculations needed to support the operability assessment. In condition report 98-1340 for Unit 2 issued on March 26, 1998, the licensee had identified that a section of vent piping connected to safety injection accumulators 2 and 3 was not adequately supported. The screening for this condition considered the piping operable based on an E-mail from engineering. This E-mail documented that piping stresses in this configuration were above the design Code (B31.1) allowable values, but below interim acceptance criteria established in the mechanical design guideline 5700-11. The licensee initiated a design change to correct this condition (not implemented as of the conclusion of this inspection period). However, as of April 23, 1999, the licensee had not provided the calculations which were relied on to support the engineering conclusions of stress levels in the piping system. Based on inspectors' discussions with the cognizant engineer, the licensee had performed calculations, but had not provided these with the operability determination, nor had the calculations been reviewed and approved as safety related calculations. Inspectors considered this example to demonstrate poor engineering practice and a lapse in the technical quality of the engineering documentation used to support this operability evaluation.

The inspectors identified a that a timely engineering response had not been provided for concerns raised in condition report 99-2448 pertaining to the alignment of a nonseismic purification system to the refueling water storage tank (RWST). In condition report 99-2448 issued on February 11, 1999, the licensee identified that the four reasons used to justify the operability of the RWST during a seismic event (as documented in Engineering Technical Direction Memorandum 99-005) were weak. The specific concerns related to the lack of calculations to support operability and that "a number of issues seem to have been overlooked in the safety screening (i.e. flooding)." The operability screening performed on February 12, 1999, questioned if the RWSTs were operable when tied to the refueling water purification system. The operations representative initially requested additional engineering details to resolve this operability question by April 1, 1999. As of the completion of this inspection period the engineering department had not provided additional evaluations to resolve this operability question. The procedural guidance for the timeliness goal in resolution of operability questions was 45 days which had not been met. Further, the licensee had a significant backlog of these types of operability questions (in excess of 500) and in general was not meeting the timeliness goal of 45 days. This backlog was independent of and in addition to the backlog of open operability determinations. The poor control of the backlog of operability determinations and questions and lack of meeting timeliness goals had been identified by the performance assurance department in condition report 99-8511 issued on April 15, 1999. The inspector noted that the licensee had taken action to establish a team to review operability determinations to ensure timely operability screenings and was planning actions to prioritize the items in the existing operability backlogs.

c. Conclusions

The inspectors identified several examples of operability determinations that contained poor quality engineering assessments. These examples combined with the large backlog (in excess of 500 items) of operability questions which have not received timely engineering resolution, demonstrated poor engineering support for the operability determination process.

E2.2 Inadequate Operability Determinations

a. Inspection Scope (40500)

The inspector reviewed a sample of operability determination screenings to evaluate the quality of these screenings.

b. Observations and Findings

In general the operability determination screenings reviewed by the inspector were in compliance with the licensee procedure PMP 7030.OPR.001 “Operability Determinations.” However, the inspectors identified two examples where licensee staff failed to follow procedure requirements for operability concerns associated with safety-related equipment documented in condition reports. These examples illustrated a lapse in rigor for the licensee’s staff performance of operability determination screenings.

In condition report 98-6391 issued on November 2, 1998, the licensee identified an operability concern associated with the Unit 1 control air to the pressurizer power operated relief valves and the pressurizer instrument lines. Portions of the air lines and instrument lines were run in the direct path of a loss of coolant accident (LOCA) blowdown and could be affected by the event. The licensee performed a screening for this condition on November 2, 1998, however this screening only addressed the concerns for the air lines and did not evaluate the pressurizer instrument lines. The inspectors’ questions, prompted the licensee to document the failure to review this issue for operability in condition report 99-8000. The lack of identification and evaluation of the pressurizer instrument lines in this screening is contrary to requirements of paragraph 4.4.5 and 4.4.6 of PMP 7030.OPR.001, Revision 0. These procedural paragraphs required identification and evaluation of the affected components in the operability determination. 10 CFR 50 Appendix B Criterion V “Instructions, Procedures and Drawings” required in part that activities affecting quality shall be accomplished in accordance with the instructions and procedures. Failure to follow the operability determination procedural requirements on November 2, 1998 for the Unit 1 pressurizer instrument lines is a Violation of 10 CFR 50 Appendix B Criterion V. This Severity Level IV violation is being treated as a Non-Cited Violation (NCV). Appendix C of the Enforcement Policy requires that for Severity Level IV violations to be dispositioned as NCVs, they be appropriately placed in a licensee corrective action program. Implicit in that requirement is that the corrective action program be fully acceptable. The adequacy of D.C. Cook's corrective action program is of concern to both the NRC and Indiana Michigan Power. Because improving the corrective action program to a satisfactory status is an integral part of Indiana Michigan Power's "Restart Plan" and is under the formal oversight of the NRC through the NRC Manual Chapter 0350 Process, "Staff Guidelines for Restart Approval," this issue will be dispositioned as an NCV (NCV 50-315/99009/03(DRS)).

In condition report 99-6255 issued on March 22, 1999, the licensee identified that piping to the both Units emergency diesel air after-coolers was potentially susceptible to erosion and that no erosion inspections of this piping had been done. The licensee performed a screening for this condition on March 23, 1999, however this screening incorrectly identified that no safety related equipment was affected. The inspectors' questions, prompted the licensee to document the failure to review the emergency diesel generators for operability in condition report 99-8008. The lack of identification and evaluation of the emergency diesel generator operability in this screening is contrary to requirements of paragraph 7.1 of PMP 7030.OPR.001, Revision 2. This procedural paragraph required review and evaluation of the affected components in the operability determination. 10 CFR 50 Appendix B Criterion V "Instructions, Procedures and Drawings" required in part that activities affecting quality shall be accomplished in accordance with the instructions and procedures. Failure to follow the operability determination procedural requirements on March 23, 1999 for the review and evaluation of the concern identified associated with the emergency diesel generators a Violation of 10 CFR 50 Appendix B Criterion V. This Severity Level IV violation is being treated as a Non-Cited Violation (NCV). Appendix C of the Enforcement Policy requires that for Severity Level IV violations to be dispositioned as NCVs, they be appropriately placed in a licensee corrective action program. Implicit in that requirement is that the corrective action program be fully acceptable. The adequacy of D.C. Cook's corrective action program is of concern to both the NRC and Indiana Michigan Power. Because improving the corrective action program to a satisfactory status is an integral part of Indiana Michigan Power's "Restart Plan" and is under the formal oversight of the NRC through the NRC Manual Chapter 0350 Process, "Staff Guidelines for Restart Approval," this issue will be dispositioned as an NCV (NCV 50-315/99009-04(DRS); NCV 50-316/99009-04(DRS)).

c. <u>Conclusions</u>

In general the operability determination screenings reviewed by the inspector were adequate, however, the inspectors identified two examples where operability determination screenings of safety related equipment had not been adequately performed. These issues were considered examples of Non-Cited Violations of 10 CFR 50 Appendix B, Criterion V and indicated a lapse in rigor for the performance of operability determination screenings.

E3 Engineering Procedures and Documentation

E3.1 Deficiencies in System Indexed Database System (SIDS)

a. Inspection Scope (40500 and 37550)

The inspectors reviewed condition reports 99-3097 and 99-4710 associated with deficiencies in the SIDS and evaluated the impact of these deficiencies on the ESRR process.

b. Observations and Findings

The SIDS database was developed to collect plant system information (such as condition reports, modifications, operability evaluations, etc.) that was distributed among other plant databases, to form a single integrated data base. The SIDS represented a powerful resource tool that enabled the ESRR teams to review a single integrated list of system deficiencies, work or modifications and as such, represented an essential element for effective system reviews. In particular, this system was the primary data source used to by the plant to identify and screen all work required to be completed prior to plant restart. Thus, deficiencies affecting operation of this system have potential for wide ranging impacts on the ESRR program and plant restart efforts.

To interface with other data bases the SIDS used system designators and translation tables of system designators to assure that transferred information was assigned to the correct ESRR system. The SIDS used system designators that were consistent with the ESRR process system designators, but differed from system designators used in other plant databases. This caused a number of items to be classified under the wrong system within SIDS. Since February 1999, licensee management had recognized that the lack of common system designators could adversely impact the ESRR process. Further, the performance assurance department had written condition reports on this problem and had discussed this issue with line organization management on repeated occasions. Although line management acknowledged the concern, as of the conclusion of this inspection period, no dates or milestones had been established to resolve this issue. To recover from system designator coding of issues under the incorrect system designator would require more resources as the ESRR effort progressed, since the SIDS contained about 200,000 items and was growing at a rate of 200-300 items per day. Examples of the challenges to the ESRR effort created by the system designator translation table disparities are discussed below.

SIDS defined a system designator for the containment structure which did not exist in the other plant databases. The other databases tracked containment items using other (often generic) applicable, system designators. For example, an issue with containment pipe supports, may be tracked under a generic “pipe support” designator with the reference to containment being in the problem description. This meant that all containment problems documented in the other databases (including those identified by the ESRR teams) were potentially misclassifed in the SIDS. For example, as of April 6, 1999, the SIDS listed only 14 items under containment, although significantly more items had been identified by the ESRR teams. Further, condition reports associated with the containment system, were being entered under the general system designator "BLDG-PLANT STRUCTURES" in the electronic corrective action data base to facilitate tracking in the SIDS. To "work around" this problem ESRR teams were performing laborious, word searches in the SIDS to identify any misclassifed items. During an April 8, 1999, weekly ESRR status meeting, the inspectors observed that several System Mangers had stated that the failure to correct this problem was impacting the efficiency of their system reviews.

The licensee had identified examples of condition reports that had been miscoded in the source database and were consequently, assigned to the wrong system in the SIDS. Of particular concern, were those reports that were considered closed and were not correctly assigned to those level 1 systems included in the ESRR scope. These condition reports could potentially fall outside of the SIDS. For example, closed report no. 94-1903, described a problem with the Unit 2 control room instrumentation distribution system, which was an ESRR level 1 system. However, this report had been classified under the control room annunciation system, which was considered a level 2 system. To correct for this potential bypass, the licensee was currently loading closed condition reports on level 2 systems into SIDS to ensure these items were not missed during the ESRR efforts.

c. Conclusions

The SIDS represented a powerful resource tool that enabled the ESRR teams to review an single integrated list of system deficiencies, work or modifications and as such, represented an essential element for effective system reviews. However, the ESRR teams were being adversely affected by the ongoing problems created by the disparity of system designators used in SIDS and other plant data bases. Further, the lack of timely licensee corrective action to resolve this problem was creating a "work around" (e.g., laborious word searches of the SIDS) effort for ESRR teams which was adversely impacting review efficiency.

E7 Quality Assurance in Engineering Activities

E7.1 System Readiness Review Board Review and Approval of ESRR Reports

a. Inspection Scope (40500)

The inspectors attended the ESRR report presentation for SRRB review, comment and approval held on April 20, 1999, on the auxiliary building ventilation systems including the engineered safety features ventilation (AES). The inspection consisted of observing this SRRB meeting, reviewing the AES ESRR report and interviewing the SRRB and ESRR team members.

b. Observations and Findings

The SRRB is responsible for performing management oversight and assessment of the ESRR Program and as such conducts a review and approval of the Phase 1 ESRR system reports in accordance with procedure PMP 7200.RST.004 "Expanded System

Readiness Review Program." The SRRB board review and approval marks the final step in the discovery phase (Phase 1) of the ESRR process. To provide effective use of the time and SRRB resources, the board members divided review responsibilities into four areas. Different members of the board were responsible for review of the safety and accident mitigation functions of the system, the SIDS items associated with the system under review, an independent system walkdown and a review of the ESRR system assessments. This structured review process offered a consistent approach for effectively reviewing the large amount of information contained in a typical ESRR system report.

During the SRRB meeting to review, discuss, comment and approve the report for the AES, the inspectors observed a summary of the AES ESRR report and significant findings presented by the team manager. This presentation demonstrated the team manager's thorough understanding of the system under review and of the relative significance of the system findings and assessments. Some of the more significant system findings identified by the ESRR team included: postulated high energy line breaks in the RHR rooms that could travel through the ventilation system and potentially impact other safety related equipment rooms, errors in ventilation related temperature calculations for the auxiliary building, lack of missile protection between trains and single failure vulnerabilities for the safety related ventilation dampers. The team concluded that there was insufficient assurance that the AES system was capable of meeting is safety and accident mitigation function for temperature control of the auxiliary building in its present configuration. This conclusion was based on the significant errors that were identified in the ventilation related temperature calculations for the auxiliary building, combined with the small margin that previously existed between calculated results and design requirements, such that there was little assurance of adequate auxiliary building room cooling for safety related equipment. On April 20, 1999, the licensee notified the NRC (pursuant to reporting requirements of 10 CFR 50.72(b)(2)(iii)(D)) for a condition that could have prevented the fulfillment of the safety function of a system needed to mitigate the consequences of an accident.

During the SRRB meeting to review, discuss, comment and approve the report for the AES, the inspectors observed that a proper SRRB membership quorum was established in accordance with the SRRB charter (Attachment 1 to the ESRR procedure). In its reviews, the SRRB enforced technical rigor, consistency of approach, and identified items requiring further ESRR team followup. Performance assurance department personnel performed surveillance monitoring/observation of the SRRB AES meeting. The Vice President of Engineering attended portions of this five hour meeting (extended and reconvened the same day). The Vice President of Engineering engaged in the commenting process on the AES report and directed that action items be assigned to the ESRR team reviewing the fire protection system for technical questions pertaining to the AES ventilation fire dampers which were outside the review scope for the AES team. This action appeared to prompt the SRRB members to begin assigning actions to technical issues/questions which had initially been considered outside the scope of the system under review. This change in SRRB activity demonstrated a positive management oversight influence in support of the ESRR process implementation.

Inspectors noted that the AES team had been assigned an action to review the historical design and licensing basis notebooks (referred to by licensee personnel as “deliverable 2") which had not yet been provided to the team. The inspectors were concerned that performing significant additional licensing basis reviews (see Section E1.2) following the SRRB approval process could change the system assessments and serve to “bypass” the SRRB review process step. The licensee managers subsequently suspended the systems scheduled for SRRB review and approval pending the receipt and review of the deliverable 2 notebooks by the ESRR teams. Inspectors considered this action to demonstrate a prompt effective management response for this issue.

c. Conclusions

The SRRB members followed a structured review process that offered a consistent approach for effectively reviewing the large amount of information contained in a typical ESRR system report. The AES ESRR report presentation demonstrated the team manager’s thorough understanding of the system under review and of the relative significance of the system findings and assessments.

The Engineering Vice President prompted the SRRB members to begin assigning actions to technical issues/questions which had initially been considered outside the scope of the system under review. This change in SRRB activity demonstrated a positive management oversight influence in support of the ESRR process implementation.

The inspectors were concerned that performing significant additional licensing basis reviews (e.g., review of the historical design and licensing basis notebooks) following the SRRB approval process could change the system assessments and serve to “bypass” the SRRB review process step. The licensee subsequently suspended the systems scheduled for SRRB review and approval pending the receipt and review of the historical design and licensing basis notebooks by the ESRR teams. Inspectors considered this action to demonstrate a prompt effective management response to this concern.

V. Management Meetings

X1 Exit Meeting Summary

The inspector presented the inspection results to members of licensee management at the conclusion of the inspection on April 23, 1999. The licensee acknowledged the inspection conclusions presented and did not identify any potential report material as proprietary.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

D. Cooper, Plant Manager
M. Finissi, Plant Engineer
D. Garner, Director Plant Engineer
S. Greenlee, Director Design Engineering
R. Huey, Performance Assurance
B. Kalinowski, Performance Assurance
W. Kropp, Performance Assurance
D. Kunsemiller, Director Regulatory Affairs
R. Powers, Senior Vice President
M. Rencheck, Vice President of Engineering
B. Sweeney
T. Taylor, Licensing
L. Thornsberry, Engineering Restart

US NRC

B. Bartlett, Senior Resident Inspector
M. Farber, Inspector

INSPECTION PROCEDURES USED

IP 40500:	Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems
IP 37550:	Engineering
IP 37700:	Design Changes and Modifications
IP 93809:	Safety System Engineering Inspection

ITEMS OPENED, CLOSED, AND DISCUSSED

<u>Opened</u>

50-315/99009-01	NCV	Failure to implement adequate design controls for the Unit 1 steam generator tube plugging and sleeving activities.
50-315/99009-02	IFI	Unreviewed change in the steam generator tube fouling factor.
50-315/99009-03	NCV	Failure to follow the operability determination procedural requirements for the Unit 1 pressurizer instrument lines.
50-315/99009-04; 50-316/99009-04	NCV	Failure to follow the operability determination procedural requirements for the concern associated with the emergency diesel generators.

<u>Closed</u>

50-315/99009-01	NCV	Failure to implement adequate design controls for the Unit 1 steam generator tube plugging and sleeving activities.
50-315/99009-03	NCV	Failure to follow the operability determination procedural requirements for the Unit 1 pressurizer instrument lines.
50-315/99009-04; 50-316/99009-04	NCV	Failure to follow the operability determination procedural requirements for the concern associated with the emergency diesel generators.

<u>Discussed</u>

None

LIST OF ACRONYMS USED

AES	Auxiliary Building Ventilation Systems including the Engineered Safety Features Ventilation
ASME	American Society of Mechanical Engineers
CFR	Code of Federal Regulations
CR	Condition Report
CRID	Control Room Instrumentation Distribution
DRS	Division of Reactor Safety
ESRR	Expanded Systems Readiness Review
FSAR	Final Safety Analysis Report
LTOP	Low Temperature Over-pressure Protection System
NCV	Non Cited Violation
NRC	Nuclear Regulatory Commission
PRT	Pressurizer Relief Tank
PA	Performance Assurance
PDR	Public Document Room
RCS	Reactor Coolant System
RHR	Residual Heat Removal System
RWST	Refueling Water Storage Tank
SIDS	Systems Indexed Database System
SRRB	System Readiness Review Board
UFSAR	Updated Final Safety Analysis Report

PARTIAL LIST OF DOCUMENTS REVIEWED

Procedures:

PMP 7030.OPR.001, Revision 0 and 2	“Operability Determinations,”
PMP 7200.RST.004,Revision (0,1,2 and 3)	“Expanded System Readiness Review Program,”

Design Changes/Modification Packages:

12-MM-282	Replace backdraft dampers on discharge of fans 1-HV-AES-1, 1-HV-AES-2, 2-HV-AES-1, and 2-HV-AES-2.
12-MM-1C-14	Remove the charcoal filter bed high temperature interlock contact from the control circuits of the control room and ESS HVAC fans.
DCP-01-826	Modify Control Air Supply to Pressurizer Power Operated Relief Valves 1-NRV-152 and 1-NRV-153.
MM-01-524	Remove Tube Lane Blocking Device From Each Unit 1 Steam Generator. Reinstall After Pressure Cleaning.
MM-12-022	Replaced Required Valves in the RCS/PR System.
RFC-01-3041	Relocate Auxiliary Pressurizer Spray 1-GRC-R506
RFC-01-3070	Provide Positive Shutoff to the Pressurizer Pressurizer Power Operated Relief Valves Backup Air Bottle and Add Test Connection.

Calculations, Vendor Analysis :

TH-97-08, Revision 0	“Reactor Coolant System Inventory After Station Blackout.”
TH-89-08, Revision 0	“Reactor Coolant System Inventory After Station Blackout.”
WCAP-14286, Revision 0	“American Electric Power Service Corporation Donald C. Cook Nuclear Plant Unit 1 Steam Generator Tube Plugging Program Engineering Report”
WCAP-14285, Revision 1	“Donald C. Cook Nuclear Plant Unit 1 Steam Generator Tube Plugging Program Licensing Report.”
WCAP-11902, Revision 0	“Reduced Temperature and Pressure Operation for the Donald C. Cook Nuclear Plant Unit 1, Licensing Report.”
WCAP-12623, Revision 0	“American Electric Power D.C. Cook Unit 1, Steam Generator Sleeving Report (Mechanical Sleeves).”

Equipment Specifications, Technical Reports and Manuals:

G-677164, Revision 0	Equipment Specification - Reactor Coolant System Series “51" Steam Generator.
406A74, Revision 4	Design Specification - 51F Replacement Steam Generator Lower Assy. & Upper Assy. Modifications.
AEP-17, Revision 0	Field Service Report- American Electric Power Indiana and Michigan Power D.C. Cook Unit 1, Multi-Frequency Eddy Current, Plug Removal, Tube Section Removal, HEJ Sleeving, Tube Stabilization, and Mechanical Plugging.

WTD-PWEE 70-31,Revision 0	Thermal Hydraulic Report - "51 Series Steam Generator Thermal Hydraulic Data Report For Donald C. Cook Power Station #1 & #2"
VTD-WEST-0409, Revision 0	"Instructions for Vertical Steam Generators for American Electric Power Services Corporation Donald C. Cook Nuclear Plant Bridgman Michigan"

<u>ESRR System Assessments-RCS/RCS Pressure Relief</u>:

DWEA001	Administrative Controls for LTOP
DWEA002	Design Basis for LTOP Settings
DWEA003	Corrective Action Program for LTOP-Related Issues
DWEA006	Technical Specification Limiting Condition for Operation-Pressurizer Code Safety Valves
DWEA007	Setpoint, Design Capacity and System Configuration For Pressurizer Code Safety Valves
DWEA008	Performance Assurance Audits
DWEA009	Operator Work-Arounds and Watch List (attribute RCS01)
DWEA010	Operator Work-Arounds and Watch List Items (attribute RCS05)
DWEA011	Operator Work-Arounds and Watch List Items (attribute RCS10)
DWEA012	Operator Work-Arounds and Watch List Items (attribute RCS11)
DWEA024	Review of RCP Maintenance Procedures and Vendor Technical Information
DWEA026	Training Lesson Plan
DWEA033	Required Reactor Coolant Inventory to Support Station Blackout
DWEA066	Unit 1 Steam Generator Plugging
HCOU001	Reactor Vessel Material Irradiation Surveillance Schedule
JMIN001	Pressure Relief Tank Internal Inspection of Coating
JMIN002	Pressure Relief Tank Design Function
JMIN003	Mechanical and Structural Calculations/Analysis
JMIN004	Vendor Technical Information (attribute RCS09)
JMIN005	Corrective Action

<u>ESSR System Assessments-120 VAC Vital Buses and CRID Inverter:</u>

DRYA004	Review and verification of the facility data base component safety classification
DRYA012	Review of training lesson plan
DRYA015	Impact of Y2K on the CRID system
DRYA016	Review of closed NRC commitments
DRYA020	SIDS review of as-built packages
DRYA021	Review of open job orders in SIDS
DRYA024	Review of change documents and temporary modifications in SIDS
DRYA026	Review of FSAR changes in SIDS
DRYA027	Review of operability determinations in SIDS
DRYA028	Review of procedure change requests in SIDS
JMCC002	Review of performance assurance and other assessments
JMCC005	Review of electrical calculations/analyses to support CRID output requirements
JMCC008	Review of regulatory guide 1.97 requirements
TTUR001	High energy line break impact on 120 VAC vital bus systems for unit 1 and unit 2
TTUR003	Assessment of 120 VAC vital instrument bus system for seismic qualifications

TTUR005	Seismic qualification utilities group review of the 120 VAC vital distribution system
TTUR007	Classification of field mounted equipment powered from CRID buses
WGRA001	Assess the effect of shared systems
WGRA002	Assess the impact of inconsistent and misleading nomenclature
WGRA009	Review of safety-related designation for CRID equipment
WGRA010	Conformance with 10CFR50.2 design basis

Safety Evaluations and Technical Specification Submittals:

Safety Evaluation, Dated March 18, 1997	7/8" Manual Taper Welded Plugs for D.C. Cook Unit 1, Revision 1 dated
Safety Evaluation, Dated April 11, 1997	The After Tube Pull Configuration of Cook Nuclear Plant Unit 1 Steam Generator 12 (Framatome Document 51-1264382-01, Revision 0
51-1264382, Revision 1	Framatome - “Safety Evaluation Input for D.C. Cook 1 Tube Pull.”
AEP NRC 1207, dated May 26, 1995	Proposed Technical Specification Changes Supported By Analysis to Increase Unit 1 Steam Generator Tube Plugging Limit and Certain Proposed Changes for Unit 2 Supported by Related Analyses.
AEP NRC 1129, Dated June 27, 1990	Technical Specification Change to Allow Sleeving The Steam Generator Tubes.

Condition Report Nos:

99-08022	Lack of timely implementation of a uniform definition of plant system designators by all Cook Plant information sources has contributed to a generic problem of improper system designation in the SIDS database; (April 11, 1999).
99-03097	System designators in the SIDS database did not encompass all the designators used in the facilities building database; (February 19, 1999)
99-04710	Lack of a uniform and systematic method for identifying and defining plant systems; (March 8, 1999)
99-04936	Face dampers on the charcoal absorber sections of the 2-HV-AES-1 and 2-HV-AES-2 fans are deformed (bowed) in the direction of air flow (March 10, 1999)
99-07485	Errors in calculation no. DCCHV01AE04N (April 1, 1999)
99-03919	Exhaust ductwork entering the plant vent stack on either unit is non-seismic class I and may result in release point other than plant stack (March 1, 1999)
99-03442	R-NED 12-DCP-0049, Rev. 1 and 2 replaced redundant (series) bypass dampers for the charcoal absorber units in the auxiliary building ventilation air units with single dampers (February 23, 1999)
99-02302	backdraft damper for 2-HV-AES-2 found open with the fan de-energized and rotating backwards (February 3, 1999)

Condition Reports with Associated Operability Evaluations:

97-3645	Arc strike on tubing to valve 1-NFP-221V1

97-1855	Local concrete temperatures around 2-CPN-2, 3 and 4 were measured as 155 of and 151 of greater than the UFSAR limit of 150 of
98-6391	Control airlines for pressurizer power operated relief valves and instrument lines may be impacted by a blowdown event
98-1340	Accumulator vent valves do not appear to be adequately supported
98-5168	Steam generator power operated pilot valve failure mode is open vice closed
99-2448	Engineering Technical Direction Memorandum 99-005 is inadequate for operability and functionality
99-2455	Unit 2 residual heat removal system pumps may be experiencing cavitation
99-2466	Unit 1 residual heat removal system pumps may be experiencing cavitation
99-6255	Erosion monitoring for essential service water system is inadequate
99-6162	West centrifugal charging pump speed increaser lube oil heat exchanger outlet pressure indicator reads to high

UFSAR Sections:

Section 9.9	Auxiliary Building Ventilation System
Section 6.0	Engineered Safety Features
Section 9.1	Control Room Ventilation System

Other documents:

Initial Expanded System Readiness Report, Auxiliary Building Ventilation Systems(Units 1and2) dated 4/16/99.

DB-12-HVSR, rev. 0	Design basis document for the engineered safety features ventilation system
DB-12-HVAB, rev. 0	Design basis document for the auxiliary building ventilation system